

October 13, 2004
GO2-04-179

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U.S. Nuclear Regulatory Commission
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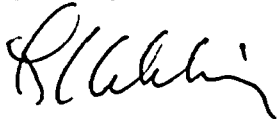
Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397
LICENSEE EVENT REPORT NOS. 2004-005-00 AND 2004-006-00**

Dear Sir or Madam:

Transmitted herewith are Licensee Event Report Nos. 2004-005-00 and 2004-006-00 for the Columbia Generating Station. These reports are submitted pursuant to 10 CFR 50.73(a)(2)(iv). The enclosed reports discuss items of reportability and corrective actions taken.

If you have any questions or require additional information, please contact Mr. DW Coleman at (509) 377-4342.

Respectfully,



RL Webring
Vice President, Nuclear Generation
Mail Drop PE04

Enclosures: Licensee Event Report 2004-005-00:
Licensee Event Report 2004-006-00

cc: BS Mallett – NRC RIV
WA Macon – NRC-NRR
INPO Records Center
NRC Sr. Resident Inspector – 988C (2)
RN Sherman – BPA/1399
TC Poindexter – Winston & Strawn
WB Jones – NRC RIV/fax

IE22

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington DC 20555-0001, or by Internet e-mail to Infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

1. FACILITY NAME

Columbia Generating Station

2. DOCKET NUMBER

05000397

3. PAGE

1 OF 3

4. TITLE

Reactor Manual Scram During Plant Startup due to High Water Level in the Pumped Drain Tank

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	15	2004	2004 - 005 - 00			10	13	2004	FACILITY NAME	DOCKET NUMBER
										05000
									FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE

Mode 1

10. POWER LEVEL

Approximately 18%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check all that apply)

<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)
<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)
<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)
<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)
<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)
<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)
<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)
<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME

Columbia Generating Station, Licensing, Fred Schill

TELEPHONE NUMBER (Include Area Code)

509 377-8599

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)☒ NO

15. EXPECTED SUBMISSION DATE

MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

During a reactor startup on August 15, 2004, with reactor power at approximately 18%, plant operators manually initiated the Reactor Protection System in response to decreasing water level in the Reactor Pressure Vessel (RPV) following a reactor feedwater pump (RFP-P-1A) trip. The steam driven RFP-P-1A tripped as designed due to high water level in the pumped drain tank. The Reactor Core Isolation Cooling system was initiated to maintain RPV level until pressure was reduced to within the capacity of the condensate booster pumps to supply water. The subsequent plant transition to mode 4 was normal in all respects and there were no safety consequences related to the event.

LICENSEE EVENT REPORT (LER)

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		2004-005-00			

17. NARRATIVE (If more space is required, use additional copies of NRC Form 366A)

Event Description

During a reactor startup on August 15, 2004, with reactor power at approximately 18%, plant operators manually initiated the Reactor Protection System (RPS) in response to decreasing water level in the Reactor Pressure Vessel (RPV) following a reactor feedwater pump (RFW-P-1A) trip. The steam driven RFW-P-1A tripped as designed due to high level in the pumped drain tank (MD-TK-1).

The Reactor Core Isolation Cooling [BN] system was used to maintain RPV level until pressure was reduced to within the capacity of the condensate booster pumps.

Immediate Corrective Action

Associated normal operating and annunciator response procedures were updated to verify normal hotwell level conditions and refrain from continued operation in an alarm state during relevant plant conditions. Operating crews were briefed on the changes.

A walkdown was conducted to identify other controllers, which were set to operate at high or low ends of the control band. During this walkdown, two offgas level indicating controllers were found to have excessive offsets. These controllers were repaired to correct this condition.

Cause

The fundamental condition that resulted in the manual scram was a condenser hotwell level controller that was adjusted to maintain level above the high-level alarm setpoint in the high end of the control band. During plant startup, hotwell level expectedly decreased when plant operators increased demand for reactor feedwater during power ascension. The hotwell level controller responded by transferring makeup water from the Condensate Storage and Transfer system (CST) [KA] to the hotwell. Considering the elevated level controller setpoint and its reduced response characteristics at this setting, the result was a greater quantity of water than is typically present in the condenser hotwell during a plant startup. Some of this excess water in the hotwell overflowed into the reactor feedwater pumped drain tank actuating a high level switch [JK] causing the feedwater pump trip.

The high hotwell level condition was exacerbated by the elevated setpoint of the hotwell level controller and a higher than normal water inventory available for transfer being stored in the CSTs. Both of these operational factors were implemented to accommodate the water management strategy implemented during the shutdown period. The adjustment of the hotwell level controller to the high end of the band was caused by inadequate interface requirements between normal operating conditions and strategies for shutdown water management.

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		2004-005-00			

The cause of this event is being investigated further and a supplemental LER will be issued if any additional insights are gained.

Further Corrective Action

Operating procedures will be revised to lower the normal operating band for CST level and administrative procedures will be revised to develop and implement improved rules and standards regarding plant water management.

Assessment of Safety Consequences

After the manual initiation of the RPS system, the subsequent operational transition to mode 4 was normal in all respects and there were no safety consequences related to the event.

Similar Events

There have been no previous similar events in which an RPS actuation resulted from actions taken to implement a water management strategy.